Proposed Steam Generator Tech Spec

REACTOR COOLANT SYSTEM (RCS)

3.4.20 Steam Generator Tube Integrity

LCO 3.4.20 Steam Generator tube integrity shall be maintained.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of LCO not met.	A.1 Be in MODE 3. AND A.2. Be in MODE 5	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.20.1	Verify steam generator tube integrity satisfies the structural integrity and accident induced LEAKAGE performance criteria in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program
SR 3.4.20.2	Verify that inspected steam generator tubes that exceed the repair criteria in the Steam Generator Program are plugged or repaired in accordance with repair methods in the Steam Generator Program.	Prior to entering MODE 4 following a SG inspection

Steam Generator Tube Integrity
Technical Specification Bases

<u>Changes from the TRM Bases are bold, blue and underlined.</u>

<u>Changes resulting from the 4/6/01 telecon with the NRC are bold, red, and underlined.</u>

Changes resulting from SGTF comments are bold, green, and underlined.

TRM Steam Generator Integrity

BASES

BACKGROUND

The three Steam Generator Performance Criteria defined by the Steam Generator Program: Accident Induced Leakage, Structural Integrity, and Operational Leakage, act together to provide reasonable assurance of tube integrity at normal and faulted conditions. Steam generator tube integrity means that the tubes are capable of performing their intended safety functions consistent with their licensing basis, including applicable regulatory requirements. The Performance Criteria and the processes required to meet them are defined by the Steam Generator Program.

The purpose of the steam generator integrity LCO is to require compliance **with the two Performance Criteria that are necessary for primary to secondary pressure boundary integrity: Accident Induced Leakage and Structural Integrity**. **These two** Performance Criteria apply to steam generator tubes and associated appurtenances (e.g. plugs, sleeves, and other repairs).

The third Performance Criterion, Operational Leakage, is addressed by the Operational Leakage Technical Specification [3.4.13].

The steam generator (SG) tubes in pressurized water reactors have a number of important safety functions. SG tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied upon to maintain the primary system's pressure and inventory. , The SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In addition, as part of the RCPB, the SG tubes are unique in that they are also relied upon as a heat transfer surface between the primary and secondary systems such that residual heat can be removed from the primary system. This steam generator tube integrity technical specification addresses the RCPB integrity function of the steam generator. The SG heat removal function is addressed by the RCS Loop Operability technical specifications.

Concerns relating to the integrity of the tubing stem from the fact that the SG tubing is subject to a variety of degradation

BASES (continued)

mechanisms. Steam generator tubes have experienced tube degradation related to corrosion phenomena, such as wastage, pitting, intergranular attack, and stress corrosion cracking, along with other mechanically induced phenomena such as denting and wear. These degradation mechanisms can impair tube integrity if they are not managed effectively.

The steam generator Performance Criteria identify the standards against which performance is to be measured. Meeting the Performance Criteria provides reasonable assurance that the steam generator tubing remains capable of fulfilling its specific safety function of maintaining RCPB integrity.

APPLICABLE SAFETY ANALYSIS

Satisfying the steam generator Performance Criteria provides reasonable assurance against tube Burst and the resulting primary to secondary leakage that might occur at normal and faulted conditions. The consequences of design basis accidents that include primary to secondary leakage are, in part, functions of the accident induced primary-to-secondary leakage rates and the dose equivalent I^{131} in the primary coolant.

The typical analysis for an event resulting in steam discharge to the atmosphere, except a steam generator tube rupture (SGTR), assumes that the total primary-to-secondary leakage from all steam generators is [1 gallon per minute] or increases to [1 gallon per minute] as a result of accident induced stresses. For accidents that do not involve fuel damage, the reactor coolant activity levels of dose equivalent I^{131} are based on the technical specification values. For accidents that do involve fuel damage, the primary coolant activity values are a function of the accident conditions.

For most PWRs, the SGTR accident is the limiting design basis event that establishes limits for these parameters. In the analysis of a SGTR event, a bounding primary-to-secondary leakage rate equal to the operational leakage rate limits in the technical specifications plus the leakage rate associated with a double-ended rupture of a single tube is assumed. The accident analysis for a SGTR assumes the contaminated secondary fluid is only briefly released to the atmosphere via safety valves and the majority is steamed to the main condenser.

BASES (continued)

For other design basis accidents such as main steam line break (MSLB), rod ejection, and reactor coolant pump locked rotor the tubes are assumed to retain their structural integrity (i.e., they are assumed not to rupture). The leakage is assumed to be at the design basis value, which is consistent with the Accident Induced Leakage Performance Criterion.

The steam generator **Accident Induced Leakage and Structural Integrity** Performance Criteria **addressed by this Technical Specification** and the limits included in the plant technical specifications for operational leakage and for dose equivalent I^{131} in primary coolant ensure the plant is operated within its analyzed condition. The dose consequences resulting from the most Limiting Design Basis Accident are within the limits defined in GDC 19 [1], 10 CFR 100 [2] or the NRC approved licensing basis (e.g., a small fraction of these limits).

Steam generator tube integrity satisfies criterion 2 of 10 CFR 50.36 (c.)(2)(ii).

LCO

The LCO requires that steam generator tube integrity be maintained. This means that the Accident Induced Leakage and Structural Integrity Performance Criteria are met. These Performance Criteria include design basis parameters that define acceptable steam generator performance.

Since conformance with the **Accident Induced Leakage and Structural Integrity** Performance Criteria can only be determined during SG inspections, compliance with the LCO during MODES 1 through 4 is determined by verifying:

- **satisfac~~tory com~~pletion of an integrity assessment during the last steam generator inspection and**
- **that plant operation is within the operating cycle defined by the Integrity assessment.**

Performance Criteria

Accident Induced Leakage and Structural Integrity are two of the three Performance Criteria defined by the Steam Generator Program. These two, along with the third Performance Criteria, Operational Leakage,

act together to provide reasonable assurance of tube integrity at normal and faulted conditions.

The NRC must approve all Performance Criteria prior to use. The required process for approval of changes to the Performance Criteria is described in Administrative Technical Specification [5.5.9]. The three Performance Criteria approved for use at [Plant] are described below.

(i) Structural Integrity Criterion

The Structural Integrity Criterion is:

"Steam Generator Tubing shall retain Structural Integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cooldown and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against Burst under Normal Steady State Full Power Operation and a safety factor of 1.4 against Burst under the Limiting Design Basis Accident~~s~~. ~~Any~~ additional loading combinations shall be included as required by existing design and licensing basis.."

Steam Generator Tubing refers to the entire length of the tube, including the tube wall and any repairs made to it, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube.

In the context of the structural integrity criterion, the Limiting Design Basis Accident is the accident that results in the largest ~~loads imposed on the~~ steam generator tubes.

The Structural Integrity Criterion can be broken into two separate considerations:

- Providing a margin of safety against tube Burst under normal and accident conditions, and
- Ensuring Structural Integrity (preventing yield or Burst)

of the SG tubes under all anticipated transients included in the design specification.

Tube Burst

Tube Burst is defined as the gross structural failure of the tube wall. The condition typically corresponds to an unstable opening displacement (e.g., opening area increased in response to constant pressure) accompanied by ductile (plastic) tearing of the tube material at the ends of the degradation.

The Structural Integrity Criterion provides reasonable assurance that a steam generator tube will not Burst during normal or postulated accident conditions. The Structural Integrity Criterion requires that the tubes not Burst when subjected to differential pressures equal to three (3) times those experienced during normal steady state full power operation and 1.4 times accident differential pressures. In addition, other loading combinations are included as required by the design and licensing basis. The safety factors of 3 and 1.4 and the requirement to include applicable design basis loads are based on ASME Code Section III subsection NB [6] requirements and Draft Regulatory Guide 1.121 [7] guidance.

For most plants the Normal Steady State Full Power Operation condition defines the most limiting parameters under which the tubes are tested. In the context of the Structural Integrity Criterion, Normal Steady State Full Power Operation is defined as the conditions existing during MODE 1 operation at the maximum steady state reactor power as defined in the design or equipment specification. Changes in design parameters such as plugging or sleeving levels, primary or secondary modifications, or T_{hot} should be assessed and their effects on differential pressure should be included if significant. Guidance on accounting for changes in these parameters is provided in the EPRI Integrity Assessment Guidelines [5].

BASES (continued)

Tube Structural Integrity

The Structural Integrity Criterion requires ~~verifies~~ that the primary pressure stresses do not exceed the yield strength for the full range of normal operating conditions including startup, operation in the power range, hot standby, cooldown, and all anticipated transients included in the design specification. All appropriate loads contributing to combined primary plus secondary stress are evaluated so as to ensure that these loads do not significantly reduce the Burst pressure for the full range of normal operating conditions including postulated accidents. For example, axial loads due to tube-to-shell temperature differences in once-through steam generator designs during postulated MSLB, or axial loading associated with locked tube supports in recirculating steam generator designs are addressed to ensure that the test conditions are at least as severe as those expected during operating and accident events.

(ii) Accident Induced Leakage Criterion

The Accident Induced Leakage Criterion is:

"The primary to secondary Accident Induced Leakage Rate for the Limiting Design Basis Accident, other than a steam generator tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all steam generators and leakage rate for an individual steam generator. Leakage is not to exceed [1 gpm per steam generator, except for specific types of degradation at specific locations where the NRC has approved greater accident-induced leakage as part of a plant's licensing basis. Exceptions to the 1 gpm limit can be applied if approved by the NRC in conjunction with approved Alternate Repair Criteria]."

In the context of the Accident Induced Leakage Criterion:

- Accident Induced Leakage Rate means the primary-to-secondary leakage occurring during postulated accidents other than a steam generator tube rupture. This includes the primary-to-secondary leakage rate existing immediately prior to the accident plus additional primary-to-secondary leakage induced during the accident.

- For steam generator primary-to-secondary pressure boundary integrity considerations, Limiting Design Basis Accident is defined as the accident that results in the minimum margin to the applicable dose limits.

The Accident Induced Leakage Criterion can be broken down into two separate considerations:

- Meeting design basis conditions, and
- Limiting Accident Induced Leakage to less than 1 gpm per steam generator under all circumstances.

Design Basis

Primary to secondary leakage is a factor in the dose releases outside containment resulting from a Limiting Design Basis Accident. The radiological dose consequences resulting from a potential primary-to-secondary leak during postulated design basis accidents must not exceed the offsite dose limits required by 10 CFR Part 100 [2] or the control room personnel dose limits required by GDC-19 [1] or the NRC approved licensing basis.

In most cases when calculating offsite doses, the safety analysis for the Limiting Design Basis Accident, other than a steam generator tube rupture, assumes a total of [1 gpm] primary to secondary leakage as an initial condition. Plant specific assumptions for Accident Induced Leakage are defined in each licensee's licensing basis. The leakage value used in the Accident Induced Leakage Criterion must be consistent with the licensing basis.

Limiting Accident Induced Leakage to 1 gpm per SG

Probabilistic safety analysis sensitivity studies have shown that accident risk is sensitive to certain design basis parameters such as 1 gpm Accident Induced Leakage per SG. As a result, leakage greater than the design basis or 1 gpm per steam generator **(whichever is less)** is not allowed unless the NRC has approved greater leakage rates as part of an Alternate Repair Criterion.

(iii) Operational Leakage Criterion

The Operational Leakage Criterion and its associated action and surveillance requirements are contained in the RCS Operational Leakage Technical Specification. The Operational Leakage Criterion is not included in **the Steam Generator Tube Integrity Technical Specification because it is one of the forms of RCS leakage that are addressed by the RCS Operational LEAKAGE technical specification and because, unlike Structural Integrity and Accident Induced leakage, it is measurable and observable by the operator during MODES 1 through 4.** The Operational Leakage Criterion is presented below to facilitate an understanding of all of the Performance Criteria since they act together to **ensure tube integrity**.

The Operational Leakage Criterion is:

> *"The RCS operational primary-to-secondary leakage through any one steam generator shall be limited to 150 gallons per day."*

An explanation of the Operational Leakage Criterion is provided in the Basses for the Operational LEAKAGE technical specification [3.4.13].

BASES (continued)

APPLICABILITY

Steam generator tubes are designed to withstand the stresses due to differential pressures as large as 3 times those experienced under normal full power operations or 1.4 times those experienced during a Limiting Design Basis Accident. This requirement is delineated in the Structural Integrity Criterion. This magnitude of differential pressure is only possible during MODES 1, 2, 3, and 4.

RCS conditions are far less challenging in MODES 5 and 6 than during MODES 1 through 4. When the plant is shutdown, primary to secondary differential pressure is low, resulting in lower stresses and reduced potential for leakage. In addition, primary coolant activity is also low. Therefore this LCO is applicable in MODES 1 through 4 only.

ACTIONS

A. The Accident Induced Leakage and Structural Integrity Performance Criteria must be met in order to ensure tube integrity.

If there is reason to believe that the Accident Induced Leakage or Structural Integrity Performance Criteria are not being met during MODES 1 through 4, an evaluation must be performed to determine compliance. The evaluation process and corresponding acceptance criteria are defined in the Steam Generator Program.

If an operating plant determines that either Performance Criteria (i) or (ii) is not met, the reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This action reduces the factors that tend to challenge tube integrity.

The allowed Completion Times are reasonable, based on operating experience, to reach the desired plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODE 5 the pressure stresses acting on the RCPB are much lower and further deterioration is much less likely.

BASES (continued)

SURVEILLANCE REQUIREMENTS

A. During shutdown periods the steam generators will be inspected as required by the Steam Generator Program. The existence of the Steam Generator Program is required by Administrative Technical Specification [5.6.10]. NEI 97-06, Steam Generator Program Guidelines [4], and its referenced EPRI Guidelines establish the content of the Steam Generator Program.

During steam generator inspections the licensee will perform an integrity assessment of the steam generator tubes. The purpose of the integrity assessment is to ensure that the Performance Criteria have been met for the previous operating period (i.e., condition monitoring), and will continue to be met for the next period (i.e., operational assessment).

The condition monitoring assessment determines the "as found" condition of the steam generator tubes with respect to the Structural Integrity and Accident Induced Leakage Performance Criteria. **The Steam Generator Program defines the methods used to determine compliance with the Performance Criteria. Use of the Steam Generator Program ensures that the methods used to** determine tube condition with respect to **the Performance Criteria are appropriate and consistent with accepted industry practices.**

The condition of the steam generator tubes with respect to the Performance Criteria is then used to assess tube integrity and the effectiveness of the Steam Generator program. This assessment may be performed analytically or by test.

The Steam Generator Program defines the frequency of SR 3.4.X.1. The frequency is determined as part of the integrity assessment. The integrity assessment determines the length of the surveillance period by using information on existing degradations and growth rates to define a cycle length that provides reasonable assurance that the tubing will meet the Performance Criteria at the next scheduled inspection.

B. During a steam generator inspection, any tube that exceeds Steam Generator Program Repair Criteria is repaired or

BASES (continued)

removed from service by plugging. Repair Criteria are those NDE measured parameters at or beyond which a tube must be repaired using an approved Repair Method or removed from service by plugging. The tube Repair Criteria establish limits for tube degradation that provide reasonable assurance that an affected tube will meet the Performance Criteria at the next scheduled inspection by allowing for anticipated growth during the intervening time interval. Because of this allowance for growth, exceeding a tube repair criterion does not necessarily mean that steam generator tube integrity is not met.

Tube Repair Criteria are either the standard through-wall (TW) depth-based criterion (e.g., 40% TW for most plants) or other Alternate Repair Criteria (ARC) approved by the NRC such as a voltage-based repair limit per Generic Letter 95-05.

The depth based criterion, approved for use at all plants by the NRC, was established when the most frequent form of degradation was general wastage corrosion. This type of degradation structurally bounds other forms of degradation and is characterized by a volumetric loss of the tube wall. This criterion was established to allow for NDE uncertainties and growth and still provide a reasonable assurance that the affected tube would not fail in the event of an accident. "Repair / plug in detection" is considered a subset of the depth based criterion. Additional basis information is provided in Draft Regulatory Guide 1.121 [8].

In recent years, improved inspection techniques, knowledge of corrosion mechanisms, and experience have revealed additional types of tube degradation in the form of cracks in the tube wall. In some instances, a reliable method of characterizing specific types of cracks at defined locations within certain steam generator designs has been developed. In these cases, the industry has developed, and the NRC has approved Alternate Repair Criteria (ARC) to permit leaving a tube in service (as opposed to plugging) when the tube has indications that fall within the limits established by the ARC. Plug or repair on detection is not an ARC.

The NRC must approve all Repair Criteria prior to use. The required process for approval of changes to the Repair

BASES (continued)

Criteria **is described in Administrative** Technical Specification **[5.5.9].** New plugging designs or methods are not ARCs and do not require prior approval by the NRC. Repair Criteria approved for use at [Plant] are:

- [40%] nominal tube wall thickness
- [Other Repair Criteria that are currently approved for use – list.]

Due to technique and analyst uncertainties, sampling.plans, and probability of detection there is a possibility that tube(s) exceeding the Repair Criteria will not be detected during a particular steam generator inspection. If the flaw(s) is detected during a subsequent inspection, the condition is not considered a reportable event unless it is determined that the Performance Criteria are not met.

Steam generator tube repairs are only performed using approved Repair Methods. Repair Methods are those means used to reestablish the RCS pressure boundary integrity of SG tubes without removing the tube from service. Plugging a steam generator tube is not a repair.

The NRC must approve all Repair Methods prior to use. The **required process** for approval of changes to the Repair Methods **is described in Administrative** Technical Specification **[5.5.9].** The Repair Methods approved for use at [Plant] are:

- [...]

During steam generator inspections, inspected steam generator tubes that exceed the Repair Criteria **are** repaired or removed from service by plugging prior to entry into MODE 4. This is necessary in order to provide reasonable assurance that tube integrity will be maintained until the next scheduled inspection.

BASES (continued)

REFERENCES

1. 10 CFR 50 Appendix A, GDC 19, *Control Room*

2. 10 CFR 100, *Reactor Site Criteria*

3. 10 CFR 50.36, *Technical Specifications*

4. NEI 97-06, *Steam Generator Program Guidelines*

5. EPRI Report TR-107621, *Steam Generator Integrity Assessment Guidelines*

6. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, *Rules for Construction of Nuclear Facility Components, Class 1 Components*

7. Draft Regulatory Guide 1.121, *Basis for Plugging Degraded Steam Generator Tubes*, August 1976

8. List applicable UFSAR sections.